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Comparison of the Dose Evaluation Methods for Criticality Accident

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The improvement of the dose evaluation method for criticality accidents is important to rationalize design of the nuclear fuel cycle facilities. The source spectrums of neutron and gamma ray of a criticality accident depend on the condition of the source, its materials, moderation, density and so on.

The comparison of the dose evaluation methods for a criticality accident is made. Some methods, which are combination of criticality calculation and shielding calculation, are proposed. Prompt neutron and gamma ray doses from nuclear criticality of some uranium systems have been evaluated as the Nuclear Criticality Slide Rule. The uranium metal source (unmoderated system) and the uranyl nitrate solution source (moderated system) in the rule are evaluated by some calculation methods, which are combinations of code and cross section library, as follows:

(a) SAS1X (ENDF/B-IV)
(b) MCNP4C (ENDF/B-VI) - ANISN (DLC23E or JSD120)
(c) MCNP4C - MCNP4C (ENDF/B-VI)

They have consisted of criticality calculation and shielding calculation.

These calculation methods are compared about the tissue absorbed dose and the spectrums at 2m from the source.

KEYWORDS: criticality accident, nuclear criticality slide rule, dose evaluation, SCALE4, MCNP4, ANISN

I. Introduction

The nuclear criticality safety designs and controls to avoid a criticality accident are one of the most important aspects in nuclear fuel cycle facilities. In addition, placements of Criticality Accident Alarm System (CAAS), reduction of personnel's exposure, and dose evaluation for general public need to be considered when a criticality accident occurs.

The simple evaluation methods, e.g. ANSI/ANS-8.3 [1] Appendix B or experimental equations, have been applied to facilities' design. The source spectrums of neutron and gamma ray of a criticality accident, however, depend on the condition of the source, its materials, moderation, density and so on. Improvement of reliability of the dose evaluation method for criticality accidents contributes to a rational design of the nuclear fuel cycle facilities. In particular, appropriate placement of CAAS is important in safety and economical point of view, because CAAS is very expensive.

Prompt neutron and gamma ray doses of some uranium criticality systems were evaluated in the Nuclear Criticality Slide Rule [2) 3)] and their results were compared with some criticality accident experiments. The rule has been one of the benchmark for the dose evaluation of criticality accident.

The objective of this study is to improve of reliability of some calculation methods through the comparison of them.

II. Outline of Nuclear Criticality Slide Rule

In 1974, a limited distribution report, entitled "A Slide Rule for Estimating Nuclear Criticality Information," was written by C. M. Hopper for Oak Ridge Y-12 plant as a tool for emergency response to nuclear criticality accident. In 1997, it was reevaluated with SCALE4 [4)] and published as "An Updated Nuclear Criticality Slide Rule."

Selected sources, relevant to five different types of nonreactor nuclear criticality accident, were as follows:

(1) Aqueous Uranyl Fluoride, ^{235}U Enrichment: 4.95wt%, $H/^{235}U$=410, Solution Density: 2.16g/cm^3
(2) Damp Uranium Dioxide, ^{235}U Enrichment: 5wt%, $H/^{235}U$ =200, Uranium Oxide Density: 3.12g/cm^3
(3) Uranyl Nitrate Solution, ^{235}U Enrichment: 93.2wt%, $H/^{235}U$ =500, Solution Density: 1.075g/cm^3
(4) Uranium Metal, ^{235}U Enrichment: 93.2wt%, Metal Density: 18.85g/cm^3
(5) Damp Uranium Dioxide, ^{235}U Enrichment: 93.2wt%, $H/^{235}U$ =10, Uranium Oxide Density: 4.15g/cm^3

In this paper, typical moderated and unmoderated source, *i.e.* (3) uranyl nitrate solution and (4) Uranium metal, were selected and calculated with some methods.

Table 1 shows number densities of source materials and air for calculation and they are the same as the report [3)].

Table 1 Number densities of source materials and air

	Uranium Metal (93.2 %)	Uranyl Nitrate Solution (93.2 %)	Air
^{235}U	4.5012E-02	1.3154E-04	-
^{238}U	2.6704E-03	9.6010E-06	-
N	-	2.8205E-04	4.00E-05
O	-	3.4012E-02	1.11E-05
H	-	6.5769E-02	-
$H/^{235}U$	0	500	-

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III. Calculation Method

1. SAS1X (SCALE4.4a) [4)]

Though this method has been the same as that of the Nuclear Criticality Slide Rule, the recalculations were executed for obtaining the spectrum because the report indicated only the prompt neutron and gamma ray doses.

The calculation flow of SAS1X is BONAMI – NITAWL – XSDRNPN – XSDRNPM – XSDOSE. The leakage spectrum from first XSDRNPM analysis is used in the second XSDRNPM shielding calculation. In this method air reflection out of detector is not considered.

The recalculation results of prompt doses were same as the graph in the report. In the original Nuclear Criticality Slide Rule, the dose unit was rads and distance in feet.

2. MCNP4C [5)]- ANISN [6)]

In this method, the leakage spectrum and the ratio from the source material were evaluated by MCNP4C-ENDF/B-IV criticality calculation at first. F1 tally (Current integrated over a surface) on the source material was used as the source in the shielding calculation. The F1 tally values were for one fission neutron and indicated the number of the leakage neutrons from source materials. On the other hand, the fission rate was obtained from the criticality calculation. The source strength for 1 fission was given as follows:

Source Strength for 1 fission = F1 Tally value / Fission Rate

Then the source was set in the small void region and the shielding calculation was executed with ANISN. The cross section libraries were DLC23E [7)] or JSD120 [8)].

3. MCNP4C - MCNP4C

Two methods, i.e. one with SSR card and the other with surface source [9)] were executed. The former was the method of the volume source with SSW card and the latter was that of the surface source made by F1 tally, for the shielding calculation.

The difference of these methods was small. The results of the method with SSR card were shown in the section IV.

4. Dose Conversion Factor

The absorbed tissue dose conversion factor in the XDC-59-8-179 [10)] was applied, the same as the Nuclear Criticality Slide Rule.

5. Mean ν and Fission Rate

The mean ν and the fission rate were obtained by the SAS1X or MCNP4C criticality calculations. They are shown in **Tables 2** and **3**, respectively. The mean ν is equal to k_{eff}/ fission rate.

Table 2 Mean ν and the fission rate for criticality calculation with SAS1X (SCALE4.4a)

	Uranium Metal (93.2%)	Uranyl Nitrate Solution (93.2%)
Mean ν	2.62	2.42
Fission Rate	3.7891E-01	4.1342E-01

Table 3 Mean ν and the fission rate for criticality calculation with MCNP4C

	Uranium Metal (93.2%)	Uranyl Nitrate Solution (93.2%)
Mean ν	2.60	2.43
Fission Rate	3.7796E-01	4.0843E-01

IV. Calculation Results

1. Uranium Metal Source

The evaluated neutron and gamma ray doses for the uranium metal source with each method are shown in **Figs. 1** and **2**, respectively. The results of ANISN-JSD120 were similar to those of ANSIN-DLC23E. The relative errors of total fluxes were less than 0.05 in MCNP4C-MCNP4C.

The difference of each method has been small nearer than 3000 cm from source center. SAS1X results have been smaller than the other results far from 3000 cm, because of the effect of the air reflection out of detector.

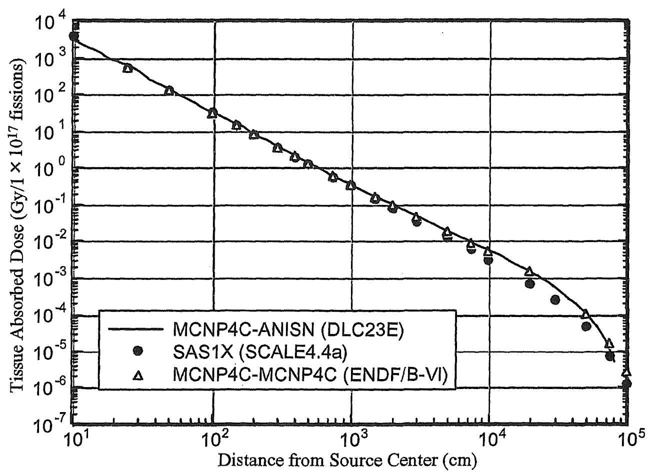

Fig. 1 Evaluated neutron dose results for uranium metal source

The n/γ ratio of the absorbed tissue dose at 2 m from source center was estimated ranging from 17.5 to 18.5. These values were large because only prompt gamma ray doses were considered. The Los Alamos accident, which included delayed gamma ray, showed that n/γ ratio was 12.0 [1)].

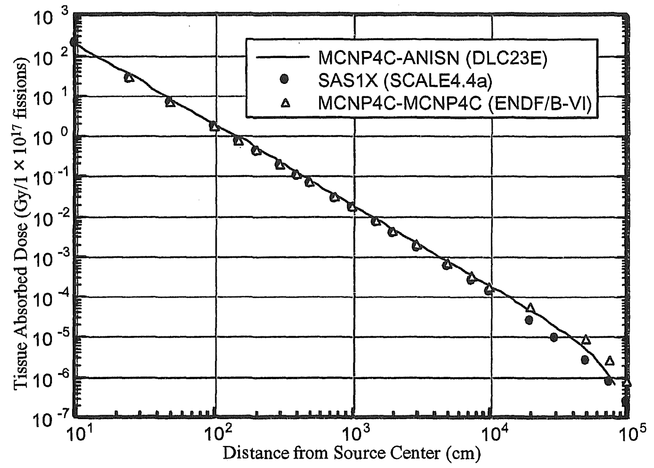

Fig. 2 Evaluated gamma ray dose results for uranium metal source

2. Uranyl Nitrate Solution Source

The evaluated neutron and gamma ray dose results for the uranyl nitrate solution source with each method are shown in Figs. 3 and 4, respectively. The results of ANISN-JSD120 were similar to those of ANSIN-DLC23E. The relative errors of total fluxes were less than 0.05 in MCNP4C-MCNP4C.

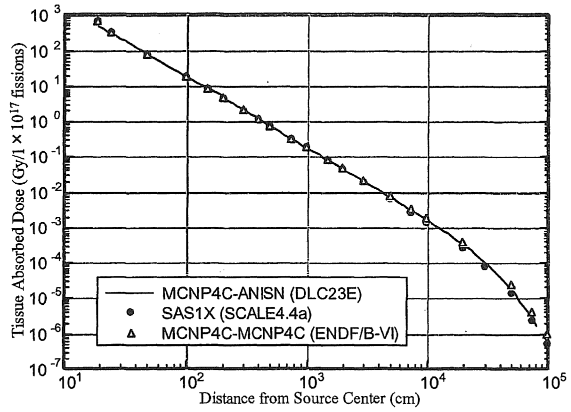

Fig. 3 Evaluated neutron dose results for uranyl nitrate solution Source.

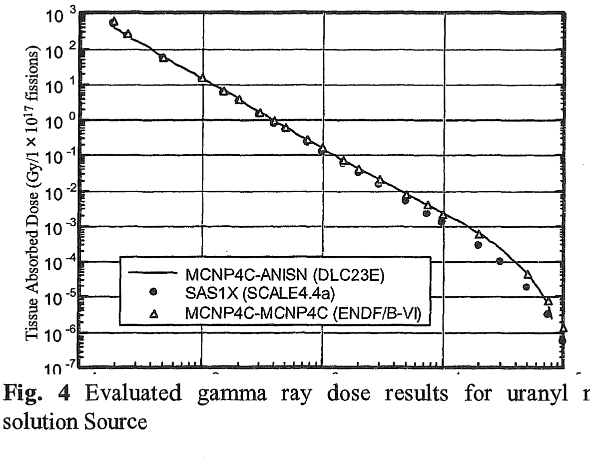

Fig. 4 Evaluated gamma ray dose results for uranyl nitrate solution Source

3. Spectrum Comparison at 2m from Source Center

The difference of each method has been small nearer than 20000 cm. SAS1X results have been a little smaller than the other results far from 20000 cm.

The spectrums at 2m from source center are compared shown in Figs. 5 to 8. They are normalized by 1 fission neutron. In case of uranium metal source SAS1X results indicate small for low energy region, because of the air reflection out of detector. These spectrums of high energy region are, however, similar in each method, so doses at 2m from source center were same.

Results of uranyl nitrate solution source with each method have been similar in the whole of energy region.

The n/γ ratio of the absorbed tissue dose at 2m from source center was estimated ranging from 0.71 to 0.79. These values were large because only prompt gamma ray doses were considered. The Y-12 accident, which included delayed gamma ray, showed that n/γ ratio was 0.3 [1].

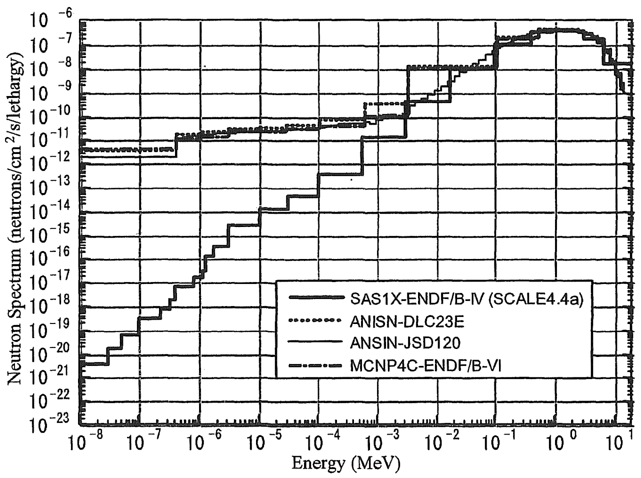

Fig. 5 Comparison of neutron spectrums for uranium metal source at 2m from source center

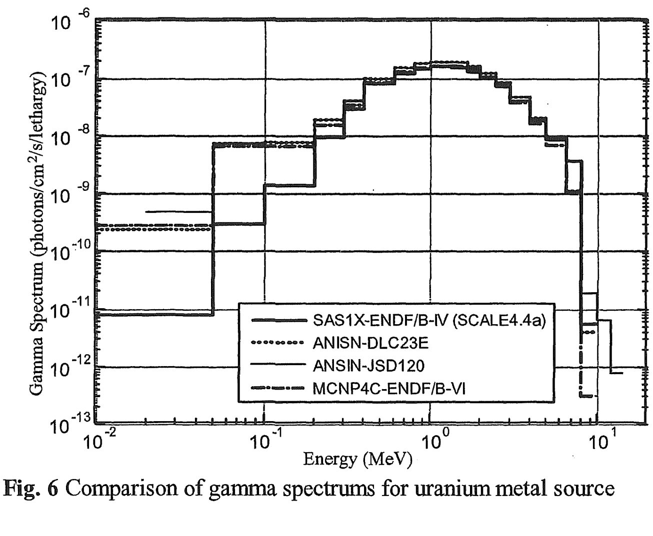

Fig. 6 Comparison of gamma spectrums for uranium metal source

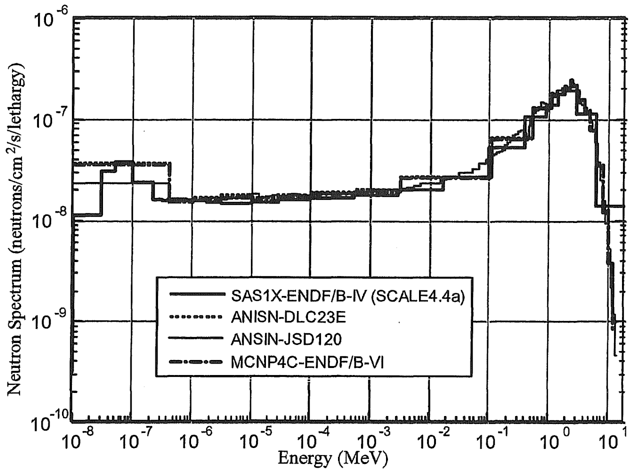

Fig. 7 Comparison of neutron spectrums for uranyl nitrate solution source at 2m from source center

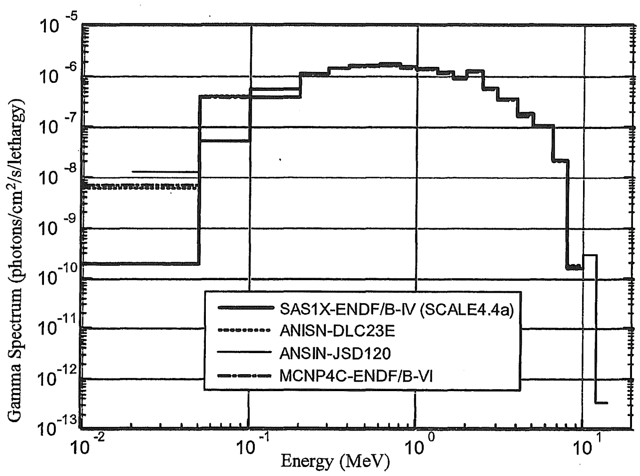

Fig. 8 Comparison of gamma spectrums for uranyl nitrate solution source at 2m from source center

V. Conclusion

These calculation methods were compared about the tissue absorbed dose and the spectrums at 2m from the source. The spectrums at the surface of the source material by each method were similar. The tissue absorbed doses of SAS1X method evaluated smaller than that of the other methods, where detection points were far from the source, because of air reflection out of the detector.

Each method can be applied to the design of the nuclear fuel cycle facilities, such as plutonium facilities, mix oxide fabrication facilities and so on, which are exposure evaluations and the CAAS placement, but the consideration of the delayed gamma ray will be necessary.

MCNP4C-MCNP4C method will be able to apply to a detail shielding evaluation.

Acknowledgement

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